NON-PUBLIC?: N
ACCESSION #: 9204170210
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Haddam Neck PAGE: 1 OF 05

DOCKET NUMBER: 05000213

TITLE: Automatic Reactor Trip During Plant Startup Due to Procedural
Inadequacy
EVENT DATE: 03/12/92 LER #: 92-009-00 REPORT DATE: 04/13/92

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 2 POWER LEVEL: 002

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On March 12, 1992, at 1618 hours, with the plant in Mode 2 and reactor power approximately 2 percent of rated thermal power, an automatic reactor trip occurred when indicated reactor power exceeded the P-7 permissive enable setpoint and plant conditions were such that the reactor protection system trip logic was satisfied. The root cause of this event was procedural inadequacy. Corrective action consisted of procedure revisions to require adjustment of indicated reactor power gains and clearly identify the P-7 enable setpoint. This event is reportable under 10CFR50.73(a)(2)(iv) since it resulted in an automatic actuation of the Reactor Protection System.

END OF ABSTRACT

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BACKGROUND INFORMATION

Nuclear overpower protection is provided by four channels of power range instrumentation of the nuclear instrumentation system (NIS) (EIIS Code: IG/JD). In addition to the reactor protection trip functions, the power range channels also provide input to the Reactor Protection System (RPS) (EIIS Code: JC) coincidentors to enable or disable certain RPS trip functions permissives, enabling or changing the required trip logic as a function of power level. One such RPS permissive is P-7, which enables, among other functions, the reactor trip function due to main steam line isolation valve (MSIV) (EIIS Code: ISV) closure, turbine stop valve (EIIS Code: ISV) closure, and loss of flow in two or more reactor coolant loops, when reactor power increases to between 7 and 8 percent of rated thermal power as indicated by 2 of 4 reactor power range instrument channels or turbine first stage pressure. The Technical Specifications Allowable Value for the P-7 enable setpoint is between 5 and 10 percent of rated thermal power (RTP) and the Trip Setpoint is between 7 and 8 percent of RTP.

During a reactor startup, the gains on each power range channel are, by procedure, set either to the maximum or so that indicated reactor power is approximately 1 percent power. This conservative practice originated at startup from the previous (Cycle 16) refueling outage, during which the nuclear instrumentation system had been replaced. It was considered that this action would provide early overpower protection in the event that the new power range channels indicated low. This conservative approach was based on past industry experience with nuclear instrumentation system replacements and erroneous indications of actual reactor power.

EVENT DESCRIPTION

On March 12, 1992, with the plant in Mode 2 and the reactor critical, operators had completed post-refueling startup physics testing and were preparing to warm the main steam lines prior to bringing the plant on-line. Normal Operating Procedure (NOP) 2.1-6, "Reactor Just Critical to Minimum Load", was commenced. A prerequisite of this procedure directed operators to branch to NOP 2.21-1, "Establishing Main Condenser Vacuum", which in turn directed operators to branch to NOP 2.19-1, "Warming Up and Pressurizing Main Steam System". At 1610 hours, with actual reactor power at approximately 0.5 percent, operators commenced a dilution of the reactor coolant system in order to partially insert control rod Bank B which had been fully withdrawn for control rod alignments. At 1618 hours an automatic reactor trip occurred. Operators carried out Emergency Operating Procedure E-0 and stabilized the plant in Mode 3. All plant systems responded normally to the trip. Actual reactor power at the time of the trip is estimated to have been approximately 2 percent.

CAUSE OF THE EVENT

The root cause of this event is procedural inadequacy in that the procedural flowpath diverted operators away from the procedural steps necessary to assure the NIS power range gains were appropriately adjusted. The automatic reactor trip was caused by indicated reactor power exceeding the P-7 setpoint with the MSIV's and the turbine valves closed, thus satisfying the reactor protection system (RPS) trip logic. Although actual reactor power increased from approximately 0.5 percent to approximately 2 percent, indicated reactor power increased from around 2 percent to approximately 7 percent. This was due to the gains on the reactor power range channels being set too high, causing a disproportionate increase in the indicated power versus actual power. While NOP 2.1-6 does contain steps requiring adjustment of the reactor power range channel gains to 1 percent until actual reactor power indicates 1 percent, these steps were preceded by a branch to another procedure. Thus, the section of NOP 2.1-6 requiring the re-adjustment of the power range gains was effectively bypassed and the needed guidance was not provided in the other procedure where an increase in reactor power is necessary to compensate for increased steam demand.

Also, as a contributing cause to this event, operating procedures did not clearly identify the P-7 trip enable setpoint. Operator training has historically emphasized the Technical Specification 10 percent Allowable Value upper limit while, in fact, the actual enable setpoint (per Technical Specification Trip Setpoint) is between 7 and 8 percent.

An additional contributing factor was the specific plant conditions at the time of the event. Prior to the automatic reactor trip, plant operators had begun admitting steam to the main steam system in the turbine building as a means of warming the steam lines and generally preparing the secondary side of the plant for operation. This process is accomplished, initially, by slowly and incrementally opening the bypass valves around the main steam line non-return valves (NRV's) (EIIS Code: ISV). This action, which is controlled by NOP 2.19-1, allows a slow warming and pressurization of the secondary side steam lines to occur. Just prior to the event, the rate of steam admission through the NRV bypass valves was such that a slight cooldown (approximately 4 degrees (F)) of the primary system had occurred, resulting in a reduction in pressurizer water level. This cooldown had been terminated prior to beginning the dilution to permit control

Bank B to be inserted further into the core. However, at the time of the trip, some residual effects of the cooldown were still in evidence in that the charging system flowrate was elevated in response to the lower than program pressurizer water level. The impact of the increased charging flow was accentuated by the charging system alignment in existence at the time. Specifically, the makeup water source at the time of the event was directly from the primary water system to the charging pump suction in lieu of the volume control tank (VCT), which is the normal source of charging system makeup. This alignment from primary wate
is utilized during post-refueling low power physics testing in order to allow for higher and more uniform dilution rates during control rod worth measurements. In this case, the alignment served, in combination with the increased charging system flowrate due to the previous cooldown, to give a significant dilution rate that, in addition to the overly conservative NIS power range gain adjustments, resulted in a rapidly increasing indicated reactor power.

SAFETY ASSESSMENT

The automatic reactor trip is reportable under 10CFR50.73(a)(2)(iv) as an event which resulted in actuation of the RPS.

The cause of the reactor trip was two of four power range NIS channels indicated power exceeding the P-7 enable setpoint while the MSIV's and a turbine stop valves were closed. Prior to the trip, one bank of control rods had been fully withdrawn for a control rod alignment to meet Technical Specification requirements for individual rod position indication, and this control rod bank was being diluted into the core in preparation for increasing power to Mode 1. Due to the power range channel gains being adjusted so that indicated power was higher than actual power, a reactor trip occurred at an indicated power level of 7 percent, while the actual core power was considerably lower. It is estimated, based on RCS loop delta-T indications, that the actual reactor power at the start of the power increase was approximately 0.5 percent, and that the automatic trip occurred at approximately 2 percent power.

The boron dilution rate at the time of the reactor trip was less than that assumed in the design basis boron dilution analysis. Thus, the reactivity insertion rate was bounded by that assumed for boron dilution and uncontrolled rod withdrawal analyses. The adjustment of the power range channel gains resulted in a reactor trip at a lower power level less than assumed in the analysis, and all systems functioned normally. In fact, the intent of adjusting the power range channel gains is to provide overpower protection at the lower power level while operating at just critical

conditions. Thus, this conservative practice resulted in automatic protective action earlier than would have otherwise been the case. Based on the above, it is concluded that this event is of low safety significance.

CORRECTIVE ACTION

Changes were made to NOP 2.1-6 to adjust the power range NIS channel gains prior to branching to another procedure. Cautions were also added to this and other procedures to clearly identify the potential of a reactor trip if P-7 clears and to identify the actual value of the P-7 setpoint. Lessons learned from this event will be included in initial and requalification licensed operator training programs.

ADDITIONAL INFORMATION

None

PREVIOUS SIMILAR EVENTS

None

CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT
RR#1 o BOX 127E o EAST HAMPTON, CT 06424-9341

April 13, 1992
Re: 10CFR50.73(a)(2)(iv)

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Reference: Facility Operating License No. DPR-61
Docket No. 50-213
Reportable Occurrence LER 50-213/92-009-00

Gentlemen:

This letter forwards the Licensee Event Report 92-009-00, required to be submitted, pursuant to the requirements of the Haddam Neck Plant's Technical Specifications.

Very truly yours,

John P. Stetz
Station Director

JPS/dl

Attachment: LER 50-213/92-009-00

cc: Mr. Thomas T. Martin
Regional Administrator, Region I
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King of Prussia, PA 19406

J. T. Shedlosky
Sr. Resident Inspector
Haddam Neck

1028-3 REV. 10/87

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